UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-390/93-42 and 50-391/93-42

Licensee: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-390 and 50-391 License Nos.: CPPR-91 and CPPR-92

Facility Name: Watts Bar 1 and 2

Inspection Conducted: June 1 through June 30, 1993

Inspector: _____ 7/23/93
G. A. Walton, Senior Resident Inspector Date Signed
Construction

Consultants: R. Compton, Nuclear Power Consultants, Inc. (paragraphs 2.a-b)
W. Marini, Pegasus, Inc. (paragraphs 2.c and 3)
D. Myers, Beckman and Associates (paragraph 3)
D. Prevatte, Powerdyne Corporation (paragraph 4)

Approved by: _____ 7/23/93
P. E. Fredrickson, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted by NRC consultants for the follow-up of 10 CFR 50.55(e) reports (construction deficiency reports), Corrective Action Tracking Documents (CATDs), Sargent and Lundy (S&L) Vertical Slice Review (VSR) Discrepancy Reports (DR), and actions on previous inspection findings.

Results:

This inspection for the closure of open items found that the licensee had adequately resolved the issues for most of the items reviewed. The quality and detail of the packages provided by the licensee for each item were adequate for inspector follow-up. One inspector follow-up item was identified pertaining to corrective action program sampling procedures (paragraph 3).

During the inspection of S&L VSR DR closeout packages, five (5) open items were identified. The items identified were: 1) an inspector follow-up item

regarding louver installation inspection documentation (paragraph 4.a); 2) an unresolved item (URI) pertaining to HVAC supply air grilles (paragraph 4.d); 3) a URI involving HVAC grille seismic integrity (paragraph 4.e); 4) a URI involving separation of cable raceways in missile zones (paragraph 4.f); and 5) a URI involving CCS containment isolation valves (paragraph 4.g).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*H. Hemmati-Arass, Mods Support Services Manager
T. Arney, Senior Quality Project Manager
M. Bellamy, Startup Manager
*K. Boyd, Site Licensing Program Administrator
*J. Chardos, Manager of Projects
*J. Christensen, Site Quality Manager
S. Crowe, Site Quality Assurance Manager
W. Elliott, Engineering Manager, Nuclear Engineering
P. Grooms, Quality Specialists Supervisor
*D. Johnson, Licensing Engineer
R. Johnson, Modifications Manager
*N. Kazanas, Vice President Completion Assurance
*D. Koehl, Technical Support Manager
*F. Koontz, Jr., Operations Engineering Manager
A. McLemore, Modifications Engineering Manager
L. Maillet, Site Support Manager
*G. Mauldin, Project Manager, Nuclear Engineering
*D. Moody, Plant Manager
W. Museler, Site Vice President
C. Nelson, Maintenance Support Superintendent
*R. Newby, Site Representative, Concerns Resolution Staff
*P. Pace, Compliance Licensing Supervisor
G. Pannell, Site Licensing Manager
*M. Singh, Manager of Projects
*V. Smith, Project Manager
K. Stinson, Project Manager
S. Tanner, Special Projects Manager
*J. Vorees, Regulatory Licensing Manager

Other licensee employees contacted included engineers, technicians, nuclear power supervisors, and construction supervisors.

NRC Consultants:
*W. Marini, Pegasus, Inc.
*D. Myers, Beckman and Associates
*D. Prevatte, Powerdyne Corporation

NRC Staff:
*G. Walton, Senior Resident Inspector, WBN

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Follow-up of Construction Deficiency Reports and Actions on Previous Inspection Findings

a. (Closed) CDR 390/86-06, 391/86-05, Additional Diesel Generator Relay Not Seismically Qualified

The subject deficiency was initially reported to the NRC in December 1985 in response to IE Notice 85-82. The licensee reported that a GE Model 12CFD differential protection relay was installed in the additional diesel generator unit; a relay that the notice had identified as potentially not seismically qualified for Class 1E service when de-energized. The licensee's review of this condition indicated that the installation of the relay at WBN was substantially different from the conditions that led to the failures noted in the IE Notice. In addition, TVA noted that the relays had been seismically qualified in the de-energized mode for the WBN configuration. NRC inspections documented in IRs 50-390, 391/86-20 and 50-390, 391/90-22 resulted in additional questions regarding the seismic qualification reports for this relay. Specific questions identified by the NRC related to Nutherm report No. 021181 were as follows:

1) What was the basis for not testing the relay in transition from energized to de-energized modes?

2) What was the basis for not testing the relay in the de-energized mode during SSE testing?

3) What was the significance of contact chatter across terminals other than 1 and 11 (the only two terminals the report highlights as having had no chatter)?

4) Did the test methodology comply with applicable requirements?

The licensee's responses to these questions were as follows:

1) The 12CFD relay operation is instantaneous, not like a timing relay. In addition, in IE Notice 85-82 the NRC "concluded that the relays should be tested in their energized and de-energized state to preclude their spurious activation during a seismic event."

2) The OBE tests performed in qualifying this relay actually enveloped the WBN revised SSE response spectra documented in SDRC Seismic Qualification Report No. 11273 dated November 15, 1982 (RIMS EEB830215001).

3) Only terminals 1 and 11 are connected to functional contacts.

4) WBN's FSAR commits to Regulatory Guide 1.100, Revision 0, which endorses IEEE 344-1975, the specification required in the TVA contract with the test vendor.

The inspector reviewed the licensee's responses and the referenced documentation. The revised WBN specific SSE response spectra was verified to be enveloped by the tested OBE spectra as reflected graphically by WBN engineering plots of the spectra, conservatively taken at elevation 773 in the additional diesel generator building. Wiring diagrams 1-45W760-82-15 (TVA) and 7013-50825-43 (Vendor) for the relay were examined to verify that only contacts 1 and 11 were functional contacts. In addition, the inspector verified that the mounting configuration of the cabinet as tested matched the actual field installation (34, 3/4 inch bolts with lockwashers). The licensee has adequately addressed the issues identified in IE Notice 85-82. This item is closed.

b. (Closed) URI 390, 391/92-45-02, QC Storage Inspection and Training Record Deficiencies

This unresolved item related to discrepancies in the documentation of QC inspections of material storage inspections and in training of inspectors to new inspection instruction QAI-10.06. Specifically, procedure SSP-10.03 and instruction QAI-10.06 both specified that the QC inspector document storage area inspections in accordance with the inspection report forms listed in Appendix B of instruction QAI-10.04. The inspection report form used by QC was Form QA9201 (Material Receipt Inspection), Appendix F of instruction QAI-10.04. However, this form only had signoff blocks for four of the required eight inspection attributes specified in the SSP and QAI. On most of the reports examined by the NRC inspector, QC had not documented on the forms that the other four attributes (M10, W02, W12, and W16) had been inspected. The NRC inspector also noted that the "closure" signatures on Form QA9201 for storage inspections were of QC inspectors, not those of a QE/QC supervisor as required by instruction QAI-10.04. The training for QC personnel on the requirements of new instruction QAI-10.06, which became effective on September 28, 1992, was not conducted until December 16, 1992. In addition, the training attendance report does not indicate the instructor, the duration, or description of the subject matter. The form had not been signed by the instructor nor was there any indication of supervisory review and approval. The inspector noted that QCIRs for at least 19 storage area inspections performed after this training date still referenced procedure SSP-10.03 as the procedure used to perform the inspection.

The licensee has revised the IR form and incorporated this form into instruction QAI-10.06. SWEC issued a letter defining the designation of signature authority for closure of IRs. The SWEC training record form has been revised to provide for documentation of the course description, the instructor's name, and duration.

SWEC also reviewed and corrected all training attendance reports to provide pertinent information that was lacking. The inspector has reviewed these actions and concludes that they correct the weaknesses identified and address the concerns identified in this unresolved item. This item is closed.

c. (Closed) VIO 390, 391/93-29-01, Inadequate and Uncontrolled Procedure

This item involves the discovery of QC personnel working to a procedure that was not controlled in accordance with site procedures and did not include or reference the necessary technical requirements. This violation was discovered by an NRC inspector while witnessing the in-situ testing of structural steel members, using a Equotip hardness tester, to determine whether the installed material met the minimum required yield strength. The inspector discovered that: (1) the procedure being used, Test Plan for Material Hardness on In-Place Structural and Miscellaneous Steel, dated February 18, 1993, had not been issued by DCRM as was required by procedure SSP-2.07, Document Control, Revision 4, and therefore was an uncontrolled document; and (2) the procedure did not include or reference the vendor manual requirements for application of appropriate correction factors to hardness readings obtained when the test was conducted with the tester in other than an upright position. It was also discovered that, although a similar procedure (Process Specification 5.M.3.1, TVA Specification for Hardness Testing of Bolts, Studs, and Other Support Components, Revision 0) existed as an approved, controlled procedure, the uncontrolled procedure was used to perform the testing.

The licensee's response to the identified violation, dated June 14, 1993, stated that normal administrative controls were not employed when performing the original in-situ testing because the intent was not to use the data collected as a primary qualification record. The testing was performed only to provide confidence that previous material control processes were adequately implemented. The licensee did agree, however, that as the data collected was included within a submittal to NRC, standard document control procedures should have been followed.

To address the identified deficiency the licensee performed the following actions:

- The uncontrolled procedure was compared to Process Specification 5.M.3.1 and it was determined that, if it was used in conjunction with the vendor manual, it would be technically equivalent to the controlled process specification. (As originally reported, the personnel performing the testing had the vendor manual in their possession and had performed the testing using the appropriate correction factors.)

- Process Specification 5.M.3.1 has been incorporated into the TVA NDE manual as Procedure N-GP-24, Hardness Testing, Revision 0, in order to make it more visible for future use.

- A memorandum (RIMS T51930614378) was issued to departmental supervisors stressing the importance of utilizing approved, controlled procedures when performing special tests and/or walkdowns.

The inspector reviewed the violation response letter and the above memorandum, and verified that the originally used procedure was technically equivalent to the process specification, and reviewed the newly issued Procedure N-GP-24, and determined that these actions are adequate to resolve the identified deficiency. This item is closed.

3. CATD Closure Package Reviews (TI 2512/15)

The inspectors reviewed CATD closure packages to determine whether the corrective actions taken resolved the associated employee concerns. The review included the associated subcategory report sections, the applicable employee concerns, the CATD, the associated corrective action plan, the closure package, and any affected hardware. With the exception of CATD 80407-WBN-01, the CATD closure packages were determined to have been adequately closed with the corrective actions properly completed. The CATDs are discussed below:

- 80406-WBN-01, Timely completion of root cause and recurrence control sections of corrective action reports

- 80214-WBN-01, QC location inspections for HVAC supports

- 80517-WBN-02, QA record status of cable insulation test cards

- 80412-WBN-01, Failure of EQC to document nonconforming conditions

- 80209-WBN-02, Alleged QC falsification of weld inspection record

- 80109-WBN-04, Inadequate inspection instruction on a Class 1E cable

- 80113-WBN-01, Failure to follow procedure SOP-QOP-03, Quality Reinspection Program, Revision 0

- 80202-WBN-01, No requirements to evaluate the quality of hardware accepted to deficient inspection procedures.

- 80208-WBN-01, Procedure QCP 3.06-3, Revision 10, appears to be incomplete.

- 80400-WBN-02, Numerous discrepancies with the dispositioning, voiding, and closure of IRNs.

- 80400-WBN-05, Lack of program to close IRNs prior to system turnover

- 80400-WBN-07, A policy existed to suppress documentation of NCRs on vendor supplied items.

- 80506-WBN-01, Uncertified inspector performed weld inspection on a support.

- 80510-WBN-01, Procedures are issued to the library and are effective prior to field personnel knowledge.

CATD 80407-WBN-01, NCRs are not using standard sampling practices for sample size and selection process for acceptability determinations, was developed from several employee concerns including concern number IN-86-243-00201 which stated "Random sampling plans (and random re-inspection plans) used to resolve NCRs might not have been done to established recognized MIL standards and therefore, sampling may not be adequate ..."

The CATD tied the concerns to a requirement in ANSI N45.2, Section 11, which states: "Where a sample is used to verify acceptability of a group of items, this sampling procedure shall be based on recognized practices and shall provide adequate justification for this sample size and selection process." The CATD corrective action plan specified three actions to resolve this issue: 1) establish a procedure to set the minimum requirements for the selection and documentation random sample processes; 2) conduct a survey of NCRs to identify those that involved a sampling process; and 3) review the identified NCRs to ensure the sampling had been appropriate.

The inspector reviewed the corrective actions and observed that the sampling procedure, WBEP 3.15 Reverification/Reinspection Sampling, had been cancelled on September 11, 1991, and was replaced by instruction EAI-8.04 of the same title. The EAI continued to describe sampling processes as developed in the earlier procedure.

The inspector reviewed the two procedures that described the adverse condition identification process, which includes NCRs, in order to verify that sampling processes were described or referenced in the related sections. The procedures were SSP-3.04, Corrective Action Program, Revision 9, and SSP-3.06, Problem Evaluation Reports, Revision 10. Both procedures referenced sampling as a means of establishing the acceptability of the extent of condition reviews. However, the sampling processes as described were not in accordance with the sampling procedures EAI-8.04.

Site procedure SSP-3.04, which describes the SCAR program, in Appendix G, Guidelines for the Extent of Condition Determination, provides a discussion of the sampling process. Similarly, procedure SSP-3.06 for the PER program, has a description of sampling in the definitions section under Extent of Condition. The one significant difference

between the SSPs and the approved sampling procedures occurred in the selection criteria of the sample population. The EAI uses a statistical approach whereas procedure SSP-3.04 states the "Amount of review is determined by the individual supervisor based on relative importance to safety." Procedure SSP-3.06 has a similar statement.

The inspector determined that neither of the procedures referenced the EAI for sampling process methods, and there was no source note to tie the discussion on sampling to the CATD. The discussion on sampling in the two SSP corrective action procedures was clearly not the same as the EAI sampling procedure on its approach to sampling.

The current procedures and sampling processes did not resolve the original employee concern. The licensee acknowledged that the two corrective action procedures did not clearly address or tie sampling requirements to the formal processes that were described elsewhere in site procedures. During the inspection period, the licensee implemented a procedure change to SSP-3.04 but had not approved the revision to SSP-3.06 to correct the identified deficiencies. These changes are to ensure that sampling processes used to establish the acceptability of nonconforming conditions would be carried out using recognized standards practices. The inspector did not determine the effect of the inconsistency of site procedures or sampling processes which had been performed. This issue is inspector follow-up item IFI 50-390, 391/93-42-01, Sampling Criteria, pending changes to SSP-3.06 and review of sampling processes as a means to justify acceptability of CAQs.

The overall current status of CATDs reviewed by NRC is as follows:

	SR CATD	TROI CLOSED	%	NRC READY	%	NRC INSP.		TOT DEFIC.		EC NOT RES.	
WBN	493	367	74%	205	41%	99		16		6	
NPS	124	103	83%	92	74%	3		0		0	
TOT	617	470	76%	297	48%	102	17%	16	16%	6	6%

4. Sargent and Lundy Vertical Slice Review Inspection

This is the first report for the Vertical Slice Review Inspection. In 1988, a Vertical Slice Review was performed by Sargent and Lundy of two systems, the component cooling system and the emergency auxiliary power system, "...to provide additional assurance that the design and construction meets licensing commitments..." The purpose of this inspection is to evaluate the VSR and the TVA responses to the VSR findings.

The scope of this evaluation includes review of the Sargent and Lundy VSR report, review of the discrepancy resolution recommendations and acceptances, review of the final discrepancy closeout packages and

supporting documentation, walkdowns, discussions with TVA people, and generation of a report. This evaluation is being performed on a sampling basis in the mechanical discipline. The discrepancy population from which the samples were drawn was as follows: The VSR generated a total of 654 items. Of these, 117 were mechanical or mechanical construction, and of these 86 had been closed out by TVA at this time. To date, 22 of these items have been addressed by the inspector, and of these, 8 were also HAAUP CAP items which have been reported separately. The following are brief summaries of the non-HAAUP CAP items which have been reviewed:

a. VSR Discrepancy Reports 8, 281, and 351, HVAC Fixed Louvers, Control building Air Intake.

DRs 8, 281, and 351 were combined since they all related to the same hardware, the control building air intake louvers.

Discrepancy 8

The control building pressurization system air intake louver is safety-related and therefore required to be designed as Seismic Category I. Contrary to this, the VSR found there were no design documents describing the mounting details and no calculations confirming the seismic integrity of the louver itself, the louver mullions, or the actual mounting details which were being employed. The VSR report concluded that this item was not design or safety significant based on an inspection which identified that the blades were welded to the mullion and licensee's engineering evaluated the weld attachment.

The inspector reviewed the closed report and found it did address the integrity of the louver installation as well as the integrity of the bird screen. It also reclassified the louver as Seismic Category I(L) (must retain its structural integrity only for a DBE). The inspector and the responsible engineer performed a walkdown of the louver, and no discrepancies were observed. The closeout was determined to be adequate.

Discrepancy 281

Part A - The VSR identified that no identification number was given to the control building air intake louver.

The VSR report concluded that this was a nondiscrepant observation based on a position that the equipment title is sufficient identification to meet the requirements of 10 CFR 50, Appendix B. The inspector determined this conclusion was proper.

Part B - The VSR identified that no mounting detail for the louver was available.

The final closeout documentation recommended that the control building air intake louver be reclassified as Seismic Category I(L) and that the mounting details shown on drawings 47W930-3, 48N1313, and 48W1259-4 are adequate. The inspector inspected the louver in the plant and concluded that the final closeout was adequate.

Discrepancy 351

The VSR identified that no documentation existed for installation inspection of the control building air intake louver.

Neither the report nor the final closeout addressed installation inspection documentation for the control building air intake louver. Therefore, the report and the closeout were incomplete.

Additional review performed by the licensee revealed two additional similar louvers for which no documentation of seismic qualifications or installation details existed. These were the north steam valve vault air intake louver and the spreading room exhaust louver. The report concluded that failure of either of these louvers would not endanger any safety-related equipment. If the spreading room exhaust louver failed, it would fall outside the control building or into its own Seismic Category I ductwork and would not affect the dampers in the system. If the north steam vault air intake louver failed, it would fall outside the auxiliary building, not affecting any safety-related equipment.

The report committed that engineering evaluations would be performed to demonstrate seismic qualification of both the louvers and their installations and that drawings would be revised to include mounting details. These proposed corrective actions were determined to be adequate.

The inspectors review of this issue found the north steam valve room air intake louver was not a louver at all, but rather a security grille with welding and mounting details on drawings 41N397-4 and 41N397-3. Additionally, inspection revealed that even if it did fail, there is no safety-related equipment upon which it could impact. For the exhaust louver originally thought to be for the spreading room, if it fails it would fall either outside the turbine building or inside its own duct. In either case, no safety-related equipment would be impacted. This component was reclassified as non-seismic. A walkdown was performed by the inspector and the responsible TVA engineer, and no discrepancies were observed. The installation inspection documentation was not addressed in the final closeout.

The VSR DR closeout package did not resolve the initial discrepancy of no documentation available for installation inspection of the louver. Additional documentation is needed in the package to address this issue. This issue is identified as

inspector follow-up item IFI 50-390/93-42-02, VSR Louver Installation Inspection Documentation, pending the licensee providing additional documentation in the closeout package to address the lack of installation inspection documentation.

b. VSR Discrepancy Report 10, HVAC Duct Insulation Fire Hazard

The VSR identified that the fire hazards associated with HVAC duct insulation in the Unit 1 auxiliary instrument room 708.0-C1 had not been identified in the latest fire hazard analysis. The combustible loading in the room was re-evaluated to include the HVAC duct insulation which added 9.75 million BTU to an existing loading of 546.75 million BTU, or 1.78 percent additional loading. Per procedure number WP-7, Fire Hazard Analysis, Revision 0, under which the analyses were performed, estimates of room combustible loading are acceptable if independent estimates do not exceed the WBN engineering estimate by more than 10 percent. Since the additional loading of the HVAC duct insulation is within the 10 percent allowance, it was deemed acceptable.

The licensee found an additional 10 percent of the rooms (nine rooms) in the auxiliary building and the control building containing insulated ducts which were re-evaluated for combustible loading. All but one, the mechanical equipment room, elevation 755.0-C1, were found to have less than 10 percent additional loading. This room contained an additional 20 percent. However, the total room loading with the addition was only 10,557 BTU. Since the room has a fixed spray system for the main source of loading, the charcoal in the control room emergency air cleanup units, and a pre-action sprinkler for the general area, this additional loading was evaluated to be insignificant.

Since the omission of the duct insulation was not found to result in inadequate fire detection or suppression systems or the exposure of safety-related systems to unacceptable fire hazards in the initial case or any of the sample cases, the report concluded that this discrepancy was not design or safety significant. A corrective action commitment was made to perform an updated walkdown and calculation before fuel loading taking into consideration the HVAC duct insulation. The proposed resolution in the report was determined to be adequate.

For the actual closeout, two new calculations were performed: 1) EPM-DOM-011190, Revision 0, dated February 8, 1990, which determined the amount of HVAC duct combustible insulation from issued TVA drawings; and 2) a new tabulation of the total combustible loading including the HVAC duct insulation and which superseded the original fire hazard analysis calculation. It was determined that, with the updated fire loadings, the plant's fire protection features are adequate for the total fixed combustible loading throughout the safety-related areas.

Contrary to the resolution report, no new walkdowns were performed. However, substitution of TVA HVAC drawings for walkdowns in determining the HVAC insulation combustible loading were determined to be acceptable. The closeout of this issue was found acceptable.

c. VSR Discrepancy Report 11, Incorrect Component Cooling System Pump Suction Valves

FSAR Section 6.3.2.2 and the CCS system description, paragraph 2.2.4, state that all CCS pumps start automatically upon receipt of safety injection and ESFAS actuation signals respectively. Contrary to the intent of these statements, the VSR found that Table 1 of the system description listed the valve positions for component cooling system pumps 1B-B and 2B-B suction isolation valves 1-FCV-70-34-B and 1-FCV-70-39B respectively as administratively locked closed. This would disable two of the five CCS pumps.

For corrective action, the licensee reviewed the CCS flow diagram and the instruction SOI-70.1 and found the documents correctly show these valves' positions as open. Section 3.1 of the system description was also found to correctly show their positions as open. Only Table 1 showed their positions incorrectly. A commitment was made in the report to correct Table 1 of the system description. All of the other CCS locked open/closed valves' positions were reviewed against the FSAR and found to be correctly stated in the system description. This discrepancy was determined by TVA to be not design or safety significant. The inspector verified that in the final closeout Table 1 of the system description was revised to show the correct locked open position for the subject valves. This closeout was found acceptable.

d. VSR Discrepancy Report 16, HVAC Supply Air Grilles

Design Criteria WB-DC-40-36.1, Classification of HVAC Systems, Revision 2, dated November 10, 1986, attachment 1, item 3, requires that grilles and balancing dampers that serve safety-related systems which must remain functional after a DBE must have the balancing dampers or adjustable louvers secured in place after balancing and that appropriate instructions and detailed drawings must be issued detailing this securing.

Contrary to these requirements, the VSR identified that the HVAC supply air grille (MK 47A380-351) with damper had adjustable vanes, but they had no physical provisions for securing them in place and no instructions or drawings detailing how they were to be secured.

The licensee performed a calculation (EPM-FM-032889, Revision 0, April 6, 1989) to show that the subject grille vanes would not close under seismic excitation. The licensee also performed a 100

percent review of the installed grilles in the control building on elevations 692.0' and 708.0' (approximately 61 grilles) and found those which were adjustable to be secured and none were of the type described above. From this review, TVA concluded that this type of grille is not prevalent in the system. Other unrelated discrepancies were also discovered in the review which were addressed by CAQRs. The licensee concluded that the discrepancy was not design or safety significant based on the results of the calculation and the review of the control building installations. Final closeout of this discrepancy by the licensee was based on acceptance of the calculation described above.

The inspector's review found both of those inputs to this conclusion appear to have been incorrect as described below. Additionally, the resolution and completion reports did not indicate what actions were to be taken to revise the design criteria described above to address this type of grille.

The calculation was based on a test where a measured weight was applied perpendicular to the thin edge of a vane and to its axis without causing it to move. The resultant moment about the axis was calculated. A moment was also calculated which would result from applying the dead weight of the vane plus the G-force from the worst case SSE acceleration applied at the same location, and this moment was compared to the test case moment. Since it was less than the test case moment, it was concluded that the vane would not move for an SSE. The inspector's review of this issue found this calculation neglected two important considerations: 1) it did not address how the vane chosen to test was representative of all the vanes on these grilles; and 2) it did not consider the forces of the air flow across the vanes. Through review of the grille vendor manual and inspection of a grille in the plant, the inspector determined that the air flow was in a direction which would tend to close the vanes. Therefore, this calculation did not demonstrate that the vanes would not change position as a result of a seismic event.

The initial resolution report concluded that, based on not finding any other similar grilles in the control building, the condition described was not prevalent. However, in walkthroughs of the auxiliary building by the inspector, other grilles of this type were found. Additionally, many of these were found with vanes missing, closed completely, or loose. (Examples include grille 1-BLD-31-3050 and 1-BLD-31-3180, auxiliary building, elevation 713'.)

Based on these walkthroughs, the inspector concluded that closeout of this DR did not encompass sufficient extent of review to provide a valid indication of the true extent of the problem. In addition, the analysis performed is also inadequate since it is based on the erroneous assumption that friction is sufficient to hold the vanes in their set position. The final closeout is also

inadequate in that it did not address revision of the design criteria to cover these type grilles.

When installed in a construction environment or in normal operation and maintenance, grilles of this design may be inadvertently damaged or moved out of adjustment. Therefore, the DR resolution in addition to showing the initial adequacy of these grilles, should address long-term measures to assure the integrity and proper adjustment of these grilles, or show that their integrity and adjustment have no safety significance. This issue is identified as unresolved item URI 50-390/93-42-03, VSR HVAC Supply Air Grilles pending licensee resolution and NRC review of the above identified problems.

e. VSR Discrepancy Report 17, HVAC Grille Seismic Integrity

Design Criteria Document WB-DC-40-36.1 requires that the method of attachment of HVAC grilles "... be analyzed to determine that it is adequate to support the item in a seismic event." Contrary to this requirement, the VSR identified that no calculations existed confirming the integrity of the mounting of HVAC grilles. The licensee performed an informal analysis and a formal calculation (WCG-E-085, Revision 0, dated May 2, 1989), and from these evaluations concluded that even with the heaviest grille installation and the worst case configuration (4 out of 10 #3 fasteners in place) seismic qualification is provided. DCNs P-2556-A, P-2557-A, and P-2558-A were issued to revise the general notes on the installation drawings to specify this minimum size fastener.

The inspectors review found several discrepancies in the analyses as follows:

1) In Calculation WCG-E-085, Sheet 7, all of the formulae contain the unstated assumption that all of the fasteners share the load evenly. This is a non-conservative assumption.

2) The informal analysis was also based on an unstated assumption that the fasteners were all evenly loaded and were symmetrically spaced about the grilles. These are non-conservative assumptions. If the fasteners are unevenly spaced, then the shear and tensile loads on them are significantly higher than calculated due to eccentric loading.

3) In the tensile stress formula, the stress area used for the fastener is different from the area used in the shear stress formula.

4) The pullout load determined throughout the calculation is based on an unstated assumption that the failure mechanism

will be by shear tear out of the sheetmetal at the threads. This is not the normal failure mechanism for screw pullout. Instead the calculation should be based on sheetmetal bending at the threads which would show a significantly lower pullout load.

5) None of the documents (VSR report, calculation, analysis) match each other in their descriptions of the grilles being analyzed. The VSR report states that the qualification is based on the largest grille which is steel and measures 38" x 48" and is qualified for four fasteners. However, in the formal calculation, the largest grille qualified is only 36" x 36" and is qualified for six fasteners. The weight of the largest grille used in the informal analysis was 33 pounds; the weight for the largest grille used in the formal analysis was 72 pounds; the actual weight of the largest grille per the formal calculation is 43 pounds.

6) The informal analysis is based on the use of 4-out-of-8 fasteners being used. The report states that it is based on 4-out-of-10 being used.

7) The informal analysis used to resolve the VSR DR should be documented in accordance with the established engineering design process to provide acceptability.

Based on the above discrepancies and deficiencies, the inspector concluded that the resolution for this DR was inadequate. This issue is identified as unresolved item URI 50-390/93-42-04, VSR HVAC Grille Seismic Integrity pending NRC review of the licensee's review and resolution of the above items.

f. VSR Discrepancy Report 22, Separation of Cable Raceways in Missile Zones

FSAR Section 8.3.1.4.2 and Design Criteria WB-DC-30-4, Section 4.2.1 required that:

> "Layout and arrangement of cable trays, conduit, wireways, etc., are such that no locally generated force or missile can destroy both redundant safety feature functions."

The VSR identified that conformance to cable tray separation requirements in missile zones could not be verified because documentation which clearly defined the missiles and their zones of influence were not available. The licensee issued two design criteria documents to address the concerns: WB-DC-40-64, Design Basis Events Design Criteria, Revision 0, to provide for the identification and evaluation of potential missiles; and WB-DC-40-65, Missiles, Revision 0, to define the potential sources of internally and externally generated missiles, the structures,

systems, and components that require missile protection, the methods used to protect against missile strike, and the acceptance criteria for missile barrier design.

As stated, the resolution/completion reports adequately addressed the stated discrepancy. In the final resolution, Calculation WBN-OSG4-191, Internally Generated Missile, Identification and Classification Study, was also issued which provided qualitative statements to show that there were no credible missile sources in most areas of the plant.

The inspector reviewed the above referenced design criteria documents and calculation and identified the following concerns:

1) In the calculation, Assumption 4.7 states, "Massive and rapid failure of equipment or component[s] to produce a missile is not credible. Technical justification: It is not credible because the material used conforms to Codes and Standards that preclude rapid failure." Based on this assumption, screening Criterion 7.3.4.K stated, "Massive and rapid failure of an equipment or component to produce a missile is not credible."

   The inspector observed that using these statements, it could be assumed that nothing in the plant could fail; therefore, there can be no missiles. The inspector maintains that these statements are too general and therefore are questionable.

   It was noted that these statements were invoked in only one location in the calculation as justification for not considering the reactor vessel seal rings as missiles. The inspector pointed out that although these statements were inappropriate justification, the seal rings could be appropriately justified as not being credible missiles based on their being entrapped by the grooves in the reactor head in which they are located.

2) In Section 4.1 of design criteria document WB-DC-40-65, Missiles, the statement is made that:

   > "Generally, a single missile shall not be capable of totally disabling any of the systems and safety functions necessary to bring the reactor to a safe shutdown condition."

   The inspector was concerned that, as given, this statement was deficient in several respects of properly stating the required criteria for missile protection as stated in 10 CFR 50, Appendix A, Criterion 4, and FSAR, Section 8.3.1.4.2. Specifically,

- The statement addressed only those systems and safety functions "...necessary to bring the reactor to a safe shutdown condition." Missing are references to those systems and safety functions necessary for "...removal of decay heat from the core..." and "...isolation of the primary containment..." as stated in the above referenced FSAR section. Although there are statements in Section 6.0 concerning decay heat removal and containment integrity, in their context, they are being applied only to barrier design and not to missile protection in general.

- The statement refers to "a single missile." It should refer to a "single event" as stated in the FSAR section referenced above. A single event could have the potential of producing more than one missile as well as other damaging conditions.

- The statement refers to a missile not "...totally disabling any of the systems and safety functions..." It could be inferred from this statement that it is permissible for a missile to partially disable a system or a safety function. However, the single failure criteria requires that in most cases a single failure must be considered in addition to failures which result from the initiating event. If the single failure were to occur in the opposite train, the plant could be left with no train to perform the required safety functions. This statement therefore does not meet the single failure criteria stated in ANSI/ANS 58.9-1981, American National Standard Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems, Sections 2 and 3.2, and IEEE Standard 379-1977, IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems, Section 5.4. The statement also does not provide for appropriate protection from missiles as required by 10 CFR 50, Appendix A, Criterion 4, Environmental and Missile Design Bases.

  Similar statements are made in Section 4.1.6 of the criteria document and the FSAR, Section 8.3.1.4.2, second paragraph. As above, these statements could be incorrectly interpreted to allow destruction of one of the redundant functions.

  Although the single failure criteria is invoked in Section 6.1.1.1.b of the criteria document, the above referenced statements are in conflict with this provision of the document, and this conflict should be resolved. Also, no specific reference is given for guidance in using the single failure criteria.

This issue is identified as unresolved item URI 50-390/93-42-05, VSR Separation of Cable Raceways in Missile Zones. pending the licensee reviewing and evaluating the calculation, design criteria document, and the FSAR to resolve the above described weaknesses and providing a specific reference for the single failure criteria application guidance

g. VSR Discrepancy Report 39, CCS Containment Isolation Valve

The VSR identified that the FSAR, Section 9.2.2.2, states that the CCS design pressure and temperature are 150 psig and 200°F respectively. The vendor drawing for CCS containment isolation check valve 1-CKV-70-692 indicated that the design pressure was 160 psig at 130°F. Therefore, this valve appeared not qualified to the FSAR requirements. This discrepancy had been previously identified by TVA in NCR 2394, and a CAQR was generated to correct this condition on this valve and other valves.

TVA's review of the NPV-1 form for the valve revealed that it was built to the Class 150 rating, and that Table NC-3512(b)-1 of the ASME Code, Section III, showed that the design conditions as described in the FSAR were within the Class 150 pressure temperature ratings. All of the other valves listed in the NCR were also found to be within Code ratings. Therefore, only documentation errors were required to be corrected. The corrective action proposed was to correct the valve nameplates, vendor drawings, NPV-1 forms and other documentation as required.

The inspectors review found that the DR resolution did not address the design requirements for this valve associated with its containment isolation safety function. Consequently, it did not attempt to discern which other valves listed on the NCR were also containment isolation valves. Per Section 6.2.1.2 of the FSAR, the design temperature for the containment is 250°F which is greater than the valve's design temperature. Therefore, the proposed resolution was incomplete.

The actual resolution of this item by TVA was to perform a calculation (EPM-LB-090889, Evaluation of Design Conditions for ASME Code Valves) which randomly sampled valves in the plant and compared their documentation with the actual design requirements. The calculation concluded that none were found where the design requirements were not enveloped by the documented design of the valves. Based on this sampling, it was then concluded that it was improbable that cases existed where there were discrepancies. It was also concluded that no documentation changes were required since the MVSR would provide traceability back to the calculation.

The inspector reviewed the calculation and found that the design temperature listed for the subject valve was 200°F. As noted above, the containment design temperature is 250°F as stated in the FSAR.

The above discrepancy had been discovered by the system engineer, and DCN 20741-A had been produced which changed the design temperature of the valve to 250°F on the flow diagrams, the bills of material, and the vendor drawings for system 70. However, it did not address the above-referenced calculation or the containment isolation valves for other systems. The inspector also reviewed another containment isolation valve, 1-FCV-67-107-B, listed in the calculation for the ERCW system. The required and supplied temperatures listed for this valve were 130°F. Therefore, this valve also does not meet the design requirements for containment.

Although the correct conditions could possibly be traced back to the calculation, there is no mechanism to assure this function occurs unless the licensee corrects the valve documentation. The inspector concluded that the closeout of this DR was inadequate.

This issue is identified as unresolved item URI 50-390/93-42-06, VSR CCS Containment Isolation Valve, pending the licensee providing a basis why the valves' documentation does not have to be complete, why a random valve selection is appropriate to bound a known documentation deficiency, and correction of the design temperature of the containment isolation valves to reflect the worst case temperature requirement based on either process or containment isolation.

h. VSR Discrepancy Report 40, Pipe Penetration Sleeve Fire Barriers

Auxiliary building pipe penetration sleeves Mk #693 and Mk #395 are required to have a 1 1/2 hour minimum fire rating. The VSR identified that no documentation existed for fire testing these sleeves. The DR resolution stated that Fire Protection Design Criteria document WB-DC-40-62, Section 3.5.3 requires that mechanical pipe sleeve seals meet the requirements of ASTM E-119, Fire Tests, and be sealed with foamed-in-place Dow Corning 3-6548 silicone RTV foam. Design criteria document WB-DC-40-66, Penetration Assemblies and Seals in Category I Structures, also requires mechanical pipe sleeve seals to meet the requirements of ASTM E-119. The DR resolution further stated that sleeve seal material, Dow Corning 3-6548 silicone RTV foam, was tested by Factory Mutual Research and shown to meet the requirements of ASTM E-119 and to provide a three-hour firestop when used in configurations similar to the subject sleeves. Dow

Corning's specification data sheet for the material also describes it as meeting the requirements of ASTM E-119 and providing a three-hour firestop. Finally, the DR resolution stated that seal sleeve drawings 47W472-2 and 47W472-3 specify that the annular space between the pipes and the sleeves for penetrations, Mark Numbers 395 and 693 respectively, be filled with eight inches of Dow Corning 3-6548 silicone RTV foam. Therefore, documentation does exist which shows that penetration sleeves Mark Numbers 395 and 693 meet and exceed the minimum fire rating of 1.5 hours. Therefore, this was determined to be a nondiscrepant observation. The inspectors review of this discrepancy found the resolution was adequate.

i. VSR Discrepancy Report 46, CCS Surge Tank Relief Valve

FSAR Section 9.2.2.3.6 stated that the CCS surge tank relief valves were sized to relieve the maximum flow rate from a tube rupture in the reactor coolant pump thermal barrier heat exchanger, and that the set pressure equaled the design pressure of the tank, 25 psig. Contrary to this, the VSR identified that the relief valve 1-RFV-70-538 did not have sufficient capacity for the described tube rupture, and the set pressure was 20 psig.

The DR further found that the CCS thermal barrier heat exchanger leakage detection design employed an arrangement which sensed differential flow between the heat exchanger inlet and outlet to automatically close the isolation valves if a differential occurred indicating significant leakage. No calculations existed which confirmed the adequacy of this design for postulated leakage conditions.

TVA determined the first item above to be non-significant since the actual relief protection for the CCS surge tank from thermal barrier heat exchanger tube failures is the differential flow sensing isolation arrangement described above. Additionally, it was determined that since the surge tank relief valve set pressure was less than the design pressure, it satisfied the ASME code requirements.

On both of these points the FSAR was incorrect, and CAQR WBT870165, Revision 1, was issued to perform a complete FSAR verification and revision to make the FSAR correctly reflect the plant design. FSAR Change Request 0881 was issued to make the FSAR agree with the actual system design and the SDD. However, review of the current FSAR revealed that the revision had not yet been made. Therefore, this item should not have been closed out.

Calculation EPM-MWL-120189, Analytical/Operational Limits for CCS Safety-Related Instruments, was generated which

determined that the recommended setpoint for the differential flow switch which affects the isolation of the thermal barrier heat exchanger upon tube rupture is 10 gpm.

The inspector reviewed the resolution of this issue and found it was adequate since the setpoint is well below the capacity of the surge tank relief valve of 187 gpm.

In the process of reviewing this discrepancy, the inspector discovered inadequacies in the testing of the check valve which separate the high pressure and low pressure CCS piping to the thermal barrier heat exchangers and in the overpressure protection of the low pressure piping. These discrepancies are identified and being tracked by NRC as URI 50-390, 391/93-40-01.

j. VSR Discrepancy Report 66, CCS Pumps Expansion Joints

The VSR identified that three new expansion joints were purchased under contract mark number 47W464-75A for the CCS pumps to replace damaged joints originally purchased under the same contract mark number without the "A" suffix. The vendor drawing for the replacement joints and the name tags on the joints had the same mark number as the original joints without the "A" suffix. TVA sampled twelve other contracts, six of which also had "A" suffixes on the mark number. No additional discrepancies were discovered between the contracts and the mark numbers. It was, therefore, concluded that this discrepancy was an isolated case. It was also concluded that this discrepancy was a non-significant documentation error. DCN P-02478-A was issued to revise the mark number to remove the suffix.

A second VSR identified discrepancy on this DR was the code N-5 data report for the CCS system piping did not list the expansion joints in Attachments C or D of the report. TVA concluded that this was a nondiscrepant observation since the code requires that only nuclear components installed in the field by welding be included in the N-5 data report. Since these expansion joints were installed by bolting, they were not required to be listed. This was confirmed by the authorized nuclear inspector. The inspectors review of the resolution and closeout of these issues determined they were adequate.

k. VSR Discrepancy Report 265, Component Cooling System Surge Tank Nozzles

The VSR identified that the component cooling system surge tank A was found to have 11 nozzles which was also reflected on the physical drawing (47W464-A, Revision 26, July 22,

1988) and the vendor drawing (N-1-2504, Revision 7, July 27, 1976). The flow diagram (47W859-1, Revision 30, 1/12/87) only showed 10 nozzles. The discrepancy report identified the missing nozzle as a 3" nozzle shown on Detail C5 of the physical drawing.

TVA determined that there were only 10 nozzles shown on the flow diagram and, actually, there were 11. However, the missing nozzle on the flow diagram was a 1" diameter x 6" long capped threaded nipple on top of the tank. The condition was determined to be nondiscrepant because TVA standards require that only those nozzles which affect the design or function of the system are required to be shown on flow diagrams, and this nozzle did neither. TVA did find that one of the nozzles was incorrectly labeled on the flow diagram. The three-inch nozzle initially identified in the DR as being the missing nozzle was mislabeled as a rupture disk when in fact it was a vacuum relief valve. DCN P-2327-B was issued to correct the flow diagram. TVA also reviewed the labeling of all the other nozzles on both the A and the B surge tanks and found no other labeling discrepancies.

The inspector reviewed the DCN and verified that the flow diagram had been corrected and by inspection verified that the valves in question were properly installed vacuum relief valves. The inspector determined that this DR was adequately closed out.

1. VSR Discrepancy Report 291, Incorrect Piping Size for RHR Heat Exchanger Relief Valves

The VSR identified that the CCS flow diagram (47W859-4) incorrectly showed the piping and valve sizes for the RHR heat exchangers 1A-A and 1B-B relief valves (1-70-551A & B, respectively) as 3/4" x 1". The correct size was 1" x 2".

The resolution description stated that DCN P-02642-A had revised the flow diagram (47W859-4) to correct the valve and line sizes and the valve locations. It further stated that the physical drawings were found to have indicated the correct size, and the actual installed piping was the correct size.

Seventeen additional valves were reviewed by TVA to determine if this condition existed elsewhere. No additional discrepancies were discovered.

The inspector reviewed the flow diagram and found that in Unit 1, although the valve locations appeared to have been revised, the line sizes had not been changed.

Further investigation revealed that the subject valves had previously been moved to the new locations by ECN 6591, and that the flow diagram had been changed to reflect this when they were revised to reflect the as-built walkdowns. The walkdowns, however, had not caught the correction required in line/valve sizes. That correction, shown on DCN P-02642, was found to be pending for the flow diagram per DCCM, but not yet incorporated. Walkdown of the valves by the inspector revealed that the installed inlet piping was actually 1.5" rather than 1". This led to the discovery that subsequent to this DCN, another DCN (M-16051) had been generated to replace these valves with larger valves which also entailed increasing the inlet line size. This revision was also pending for the flow diagram, awaiting installation closeout which was delayed awaiting delivery of the new valves. This revision superseded the revision covered by the DR.

Although, the closeout of the DR was improper, the resultant design changes corrected the questions regarding line size and location. Final closeout of the issue is pending installation of the new valves. The resolution of the issue is considered adequate.

5. Exit Interview

The inspection scope and findings were summarized on June 30, 1993, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Status	Description and Reference
390/86-06 391/86-05	Closed	CDR - Additional Diesel Generator Relay Not Seismically Qualified (paragraph 2.a)
390/92-45-02 391/92-45-02	Closed	URI - QC Storage Inspection and Training Record Deficiencies (paragraph 2.b)
390/93-29-01 391/93-29-01	Closed	VIO - Inadequate and Uncontrolled Procedure (paragraph 2.c)
390/93-42-01 391/93-42-01	Open	IFI - SSP Sampling Criteria (paragraph 3)

390/93-42-02	Open	IFI - VSR Louver Installation Inspection Documentation (paragraph 4.a)
390/93-42-03	Open	URI - VSR HVAC Supply Air Grilles (paragraph 4.d)
390/93-42-04	Open	URI - VSR HVAC Grille Seismic Integrity (paragraph 4.e)
390/93-42-05	Open	URI - VSR Separation of Cable Raceways in Missile Zones (paragraph 4.f)
390/93-42-06	Open	URI - VSR CCS Containment Isolation Valve (paragraph 4.g)

6. List of Acronyms and Initialisms

ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
BTU	British Thermal Units
CAP	Corrective Action Program
CAQR	Condition Adverse to Quality Report
CATD	Corrective Action Tracking Document
CCS	Component Cooling System
CDR	Construction Deficiency Report
CFR	Code of Federal Regulations
DBE	Design Base Event
DCCM	Document Control Change Management System
DCN	Design Change Notice
DCRM	Document Control Records Management
DR	Deficiency Report
EAI	Engineering Administrative Instruction
ECN	Engineering Change Notice
EQC	Electrical Quality Control
ERCW	Essential Raw Cooling Water
ESFAS	Engineered Safeguards Features Actuation System
FSAR	Final Safety Analysis Report
GE	General Electric Company
gpm	gallons per minute
HAAUP	Hanger Analysis and Update Program
HVAC	Heating, Ventilation, and Air Conditioning
IE	Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IR	Inspection Report
IRN	Inspection Rejection Notice
MK	Mark Number
MVSR	Master Valve Status Report
NCR	Nonconformance Report

NDE	Nondestructive Examination
NPS	Nuclear Procedures Staff
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
psig	pounds per square inch gauge
QA	Quality Assurance
QAI	Quality Administrative Instruction
QC	Quality Control
QCIR	Quality Control Inspection Report
QCP	Quality Control Procedure
QE	Quality Engineering
RHR	Residual Heat Removal
RIMS	Records Information Management System
RTV	Room Temperature Vulcanizing
SCAR	Significant Corrective Action Report
SDD	System Design Description
SDRC	Structural Dynamics Research Corporation
SOI	System Operating Instruction
SSE	Safety Shutdown Earthquake
SSP	Site Standard Practices
SWEC	Stone and Webster Engineering Corporation
TI	Technical Instruction
TROI	Tracking and Reporting of Open Items
TVA	Tennessee Valley Authority
URI	Unresolved Item
VIO	Violation
VSR	Vertical Slice Review
WBEP	Watts Bar Engineering Project
WBN	Watts Bar Nuclear Plant